March 13, 2001

Mr. J. Morris
Site General Manager
Monticello Nuclear Generating Plant
Nuclear Management Company, LLC
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-263-01-02(DRP)

Dear Mr. Morris:

On February 13, 2001, the NRC completed a baseline inspection at your Monticello Nuclear Power Plant. The enclosed report documents the inspection findings discussed on February 14, 2001, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to reactor safety, verification of performance indicators, event followup, and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified two issues of low safety significance (Green) involving three violations of NRC requirements. The violations involved instances of: (1) a failure to promptly declare systems inoperable as required by Technical Specifications combined with the inability to promptly restore the affected systems to operable, (2) a failure to make notifications as required by 10 CFR 50.72, (3) a failure to follow procedures that implement Inservice Inspection programs. These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the enclosed inspection report. The three violations are being treated as Non-Cited Violations, consistent with Section VI.A of the Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region III, 801 Warrenville Road, Lisle, Illinois 60532-4351, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors' Office at the Monticello Nuclear Generating Plant.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http:www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Bruce L. Burgess, Chief
Division Projects Branch 2

Docket No. 50-263
License No. DPR-22

Enclosure: Inspection Report 50-263-01-02(DRP)

cc w/encl: Plant Manager, Monticello
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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	50-263-01-02(DRP)
Licensee:	Nuclear Management Company, LLC
Facility:	Monticello Nuclear Power Plant
Location:	2807 West Highway 75 Monticello, MN 55362
Dates:	January 1 through February 13, 2001
Inspectors:	Stephen Burton, Senior Resident Inspector Dan Kimble, Resident Inspector Ken Riemer, Regional Inspector Katherine Green-Bates, Regional Inspector Jeff Harold, Regional Inspector Robert Jickling, Regional Inspector
Approved by:	Bruce L. Burgess, Chief Projects Branch 2 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC-licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

Monticello Nuclear Power Plant
NRC Inspection Report 50-263-01-02(DRP)

IR 05000263-01-02(DRP), on 1/1-2/13/2001; Nuclear Management Company, LLC; Monticello Nuclear Power Plant; Personnel Performance During Nonroutine Plant Evolutions and Events; Resident Operations Report.

The inspection was conducted by resident and regional inspectors. The report covers a 6½-week period of resident inspection. The inspection identified two green findings encompassing three noncited violations, and one unresolved item. The significance of all of the findings are indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

Cornerstone: Mitigating Systems

- Green. On January 19, 2001, the licensee identified that they were not in compliance with the ASME Boiler & Pressure Vessel Code, Section XI, 1986 Edition. The licensee determined that they had failed to involve the Authorized Nuclear Inservice Inspector (ANII) in repair and replacement activities for safety-related snubbers. One noncited violation was identified against Technical Specification 3.6.H.2.c for failure to take actions required by the technical specifications for inoperable snubbers. In addition, two non-cited violations were identified for failure to report via 10 CFR 50.72 and for the failure to follow procedures in accordance with Technical Specification 6.5. Later, on January 24, the licensee determined that a plant shutdown was required by Technical Specifications.

  The risk significance of this finding was determined to be very low because the licensee was able to determine through engineering evaluations that the as-found condition of the snubbers had no adverse effect on the supported components and that they would retain their structural integrity in the event of a design basis seismic event. (Section 1R14.2)

- Green. On January 29, 2001, the licensee identified that a plant shutdown was required by Technical Specifications due to non-compliance with the ASME Boiler & Pressure Vessel Code, Section XI, 1986 Edition concerning Inservice Inspection activities. Specifically, the licensee determined that they had failed to involve the Authorized Nuclear Inservice Inspector (ANII) in repair and replacement activities for reactor pressure vessel safety relief valves (SRVs) and as a result, declared all SRVs inoperable. Additionally, the licensee requested and received a Notice of Enforcement Discretion (NOED) to allow the use of operability evaluations to disposition this, and future, Code non-compliance issues associated with Inservice Inspection activities. One unresolved item was identified that will address the extent of condition for identified non-compliances with the ASME Boiler and Pressure Vessel Code, Section XI.

The risk significance was determined to be very low because the licensee was able to demonstrate, by operability evaluations, that the SRVs were able to perform their intended functions. (Section 1R14.3)

Report Details

Summary of Plant Status: The unit operated at approximately full power from the beginning of the inspection period on January 1, 2001, until January 15, 2001, when power was reduced by approximately 1 percent for 3 hours to support troubleshooting of control rod drive 06-23. Upon resumption of full power operation on January 15, 2001, the unit again operated at approximately full power until a Technical Specification required shutdown was commenced on January 25, 2001, for inoperable snubbers (Section 1R14.2). The shutdown was terminated at approximately 78 percent power and the unit returned to full power operation on January 26, 2001. Unit operation at approximately full power continued until January 30, 2001, when a second Technical Specification required shutdown was commenced for inoperable reactor vessel Code safety relief valves (Section 1R14.3). This shutdown was terminated at approximately 86 percent power and the unit returned to full power operation later on January 30, 2001. The unit operated at approximately full power from January 30, 2001, until February 10, 2001, when a routine power reduction to approximately 75 percent power was commenced to support turbine valve testing and miscellaneous other work. The unit was returned to full power operation on February 11, 2001, and remained at approximately full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignment

a. Inspection Scope

Due to the system's risk significance, the inspectors selected the Residual Heat Removal (RHR) System for a complete walkdown. The inspectors walked down the system to verify mechanical and electrical equipment lineups, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. Documents reviewed included:

- Electrical Wiring Diagram, NF-36298-1, Revision M, "Monticello Nuclear Generating Plant Electrical Load Flow One Line Diagram"

- General Electric Letter No. GLN-97-053 to Northern States Power Company dated November 17, 1997, Engineering Evaluation, "Emergency Service Water System Engineering Evaluation Task 14.01"

- NuTech Engineers Letter No. SAT-91-071 to Northern States Power Company dated April 30, 1991, "ISI [Inservice Inspection] Findings of Incomplete Bolt Thread Engagement P.O. No. PB328OMQ/E90R495"

- Design Basis Document (DBD), Section B.8.1, "Design Basis Document for Residual Heat Removal System"

- Operations Manual:
  - Section B.3.4-01, Revision 2, "Residual Heat Removal System"
  - Section B.3.4-05, Revision 13, "Torus Cooling Mode - Startup"

- Modifications:
  - Design Change 93Q200, Revision 0, "#12 RHR Motor Replacement"
  - Monticello "Jumper Bypass Evaluation for Plug Tube in RHR Heat Exchanger E-200B," dated November 1, 1999
  - Design Change 00Q120, Revision 0, "RHR Heat Exchanger Tube Plug"

- Calculations:
  - Calculation CA-93-047, Revision 1, "RHR Room Heat Up Calculation Based on Heat Loads From New #12 RHR Pump Motor"
  - Calculation CA-96-113, Revision 0, " Temperature of RHR Rooms During DBA LOCA"
  - Calculation CA-97-074, Revision 0, " RHR Room Temperature Response to USAR Rev 2, Torus Water Temperature Profile"

- Equipment Isolations:
  - 00-80491, Version 1, "2RS, 2R, 3N4, 3N5, & CLP Maintenance Isolation"

- Piping and Instrument Diagrams (P&IDs):
  - M-112, Revision BF, "[Division 2] Residual Heat Removal Service Water and Emergency Service Water Systems"
  - M-120, Revision BH, "[Division 2] Residual Heat Removal System"
  - M-121, Revision BK, "[Division 1] Residual Heat Removal System"

- Technical Specifications:
  - Section 3/4.9, "Auxiliary Electrical Systems," and Basis
  - Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis

- Technical Specification Amendment to Facility Operating License [Docket 50-263] No. 102 (Power Uprate Program)

- Updated Safety Analysis Report (USAR), Section 6.2, Revision 18, "Residual Heat Removal System"

- Condition Reports:
  - 19981642, "RHR System Secondary Mode Operation & Re-alignment"
  - 19990966, "Single Failure Vulnerability of the RHR System When in Suppression Pool Cooling Mode"
  - 20000158, "Relief Valve No. RV-1991 Fails Section XI As Found Testing"
  - 20000159, "Relief Valve No. RV-1993 Fails Section XI As Found Testing"
  - 20000187, "RV-2005 Fails Section XI As Found Testing per Work Order 9904911"
  - 20000530, "Large Number of Relief Valves Fail Section XI Testing Low Out of Expected Range"

- Work Orders:
  - 0000229, "Plug #12 RHR Heat Exchanger Tubes"
  - 9904911, "Section XI RHR Check Valve No. 2005 Test"

- Procedures:
  - 2154-12, Revision 33, "Residual Heat Removal System PreStart Valve Checklist"
  - 0255-11-III-4, Revision 24, "14 Emergency Service Water Pump and Valve Operability Tests"

b. Findings

There were no findings identified during this inspection.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) to ensure rule requirements were met for the selected systems. The following systems were selected based on their being designated as risk significant under the Maintenance Rule, or their being in the increased monitoring (Maintenance Rule category a(1)) group:

- Process Radiation Monitoring (PRM)

- Secondary Containment (SCT)

- Reactor Core Isolation Cooling (RCIC)

The inspectors verified the licensee's categorization of specific issues, including evaluation of the performance criteria. The inspectors reviewed the licensee's implementation of the maintenance rule requirements, including a review of scoping, goal-setting, and performance monitoring; short-term and long-term corrective actions; functional failure determinations associated with the condition reports listed below; and current equipment performance status. The documents reviewed included:

- NUMARC [Nuclear Management and Resources Council] 93-01, Revision 2, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

- Regulatory Guide 1.1.6, Revision 1, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

- Engineering Work Instruction 05.02.01, Revision 3, "Monticello Maintenance Rule Program Document"

- Monticello Maintenance Rule Periodic Assessment Report, 1st Quarter - 2000

- Monticello Maintenance Rule Availability Trend for Reactor Core Isolation Cooling

- Operations Manual:
  - Section B.5.11, "Process Radiation Monitoring System"
  - Section B.4.2, "Secondary Containment/Standby Gas Treatment"
  - Section B.2.3, "Reactor Core Isolation Cooling"

- Technical Specifications Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis

- Maintenance Rule Program System Basis Document:
  - Section B.5.11, Revision 2, "Process Radiation Monitoring System"
  - Section B.4.2, Revision 1, "Secondary Containment"
  - Section B.2.3, "Reactor Core Isolation Cooling"

- USAR, Revision 18:
  - Section 7.5.2, "Process Radiation Monitoring System"
  - Section 5.3, "Secondary Containment System & Reactor Building"
  - Section 10.2.5, "Reactor Core Isolation Cooling"

- Alarm Response Procedures (ARPs):
  - 259-A-2, Revision 2, "Reactor Building Ventilation Effluent HI-HI Radiation"
  - 4-B-22, Revision 3, "Drywell CAM [Continuous Air Monitor] Trouble"

- Condition Reports:
  - 19990581, "Unplanned Entry into LCO for 'B' Stack WRGM [Wide Range Gas Monitor] during Procedure 0355, 'Stack WRGM Source Check'"
  - 19990621, "Reactor Building Ventilation Wide Range Gas Monitor 'B' High Range Channel Spiking"
  - 19991107, "Low Sample Flow Alarm on Stack 'B' WRGM while Performing Surveillance Test 0356"
  - 19991802, "SJAE [Steam Jet Air Ejector] Offgas Radiation Monitor Channel 'A' has Exceeded its Maintenance Rule Reliability Criteria"
  - 19992463, "Declared 'B' Reactor Building Ventilation WRGM Inoperable due to Erratic Operation. Entered a 7-Day LCO per Technical Specification Table 3.14.1. Work Order 9906881"
  - 19992478, "Declared 'B' Reactor Building Ventilation WRGM Inoperable due to Purging Problems During Test 0016. Entered 7-Day LCO. Work Order 9906895 submitted"
  - 20000470, "While Shutting Down Y71 per B.9.13 -- 05.E.1, Revision 9, Lost Channel 'A' of Stack and Reactor Building Ventilation WRGMs to Support 13 Battery PM"
  - 20004523, "Unplanned LCO Entered as Required by Technical Specification Table 3.14.1 for Stack 'A' WRGM Sample Pump Inadvertently Shut Off"

- 20000903, "Standby Gas Treatment Room Found to be at Lower Pressure Than Reactor Building During Work Order 9908430 and Test 0151-1"
- 20001035, "8-hour LCO not Entered for Inoperable Secondary Containment Isolation Dampers when Standby Gas Treatment Trains Deenergized for Online Maintenance/Repair"
- 19993418, "Entered a 14 Day Unplanned LCO for RCIC Due to Apparent FI-13-91 Downscale"
- 20000132, "Local Leakage Rates Exceed Technical Specification Limits (2000 Refueling Outage) & Maintainence Rule Goal Not Met"

• Work Orders:
  - 9905607, "FI-13-91 RCIC Flow Indicates 30 gpm Instead of Zero"
  - 9907616, "Troubleshoot RCIC Turbine Low Flow Response"
  - 9907636, "Troubleshoot RCIC Turbine Low Flow Response"
  - 9907653, "Troubleshoot RCIC Governor Valve, Servo & Linkage"
  - 9907654, "Replace EGR for RCIC Turbine"
  - 9907657, "Investigate and Repair RCIC Governor Valve Stem"
  - 9907669, "Align RCIC Governor Valve Servo & Linkage"
  - 9907939, "RCIC Flow Indicator Downscale"
  - 9908072, "Replacement of RCIC-57 and RCIC-59"
  - 9908661, "Disassemble and Inspect YS 4262"

b. Findings

There were no findings identified during this inspection.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed and observed emergent work, preventive maintenance, or planning for risk significant maintenance activities. The inspectors observed maintenance or planning for the following activities or risk significant systems undergoing scheduled or emergent maintenance:

• Weekly Scheduling and Planning Meetings

• Daily Work Planning Update and Emergent Work Review

• Emergent Work/Repair of CV-1474, "Instrument to Service Air Low Pressure Isolation Valve"

The inspectors also reviewed the licensee's evaluation of plant risk, risk management, scheduling, and configuration control for these activities in coordination with other scheduled risk significant work. The inspectors verified that the licensee's control of activities considered assessment of baseline and cumulative risk, management of plant configuration, control of maintenance, and external impacts on risk. In-plant activities were reviewed to ensure that the risk assessment of maintenance or emergent work was complete and adequate, and that the assessment included an evaluation of external factors. Additionally, the inspectors verified that the licensee entered the appropriate risk category for the evolutions. The documents reviewed included:

- Procedures:
  - 4AWI-04.01.01, Revision 27, "General Plant Operating Activities"
  - SWI-14.01, Revision 0, "Risk Management of On-line Maintenance"

- Work Orders:
  - 0105511, "Repair CV-1474"
  - 0105453, "Investigate CV-1474 Not Fully Closing"

- Weekly Planning Meeting Primavera™ Printouts:
  - Weekly Planning Meeting Update for the week of 1/7/01
  - Weekly Planning Meeting Update for the week of 1/14/01
  - Weekly Planning Meeting Update for the week of 1/21/01
  - Weekly Planning Meeting Update for the week of 1/28/01
  - Weekly Planning Meeting Update for the week of 2/4/01
  - Weekly Planning Meeting Update for the week of 2/11/01

b. Findings

There were no findings identified during this inspection.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events

.1 Annual Evaluation of Licensee Event Reports (LERs)

a. Inspection Scope

The inspectors reviewed all of the licensee LERs written during the 2000 calendar year, focusing on those involving personnel response to non-routine conditions. Where applicable, the inspectors determined whether or not licensee personnel responded in accordance with applicable procedures and training, and evaluated the occurrence and subsequent personnel actions using the Significance Determination Process.

b. Findings

There were no findings identified during this inspection.

.2 Initiation of Technical Specification Required Shutdown Due to ASME Section XI Non-Compliance with Snubber Inspection Requirements

a. Inspection Scope

The inspectors reviewed personnel performance, recovery actions, and licensee response to the initiation of a plant shutdown required by Technical Specifications due to non-compliance with the ASME Boiler & Pressure Vessel Code, Section XI, 1986 Edition, (hereafter referred to as the Code) for safety related snubbers. To evaluate the occurrence, the inspectors reviewed operator logs, equipment records, licensee response, applicability to the significance determination process, and contingency plans. Documents reviewed included:

- Technical Specifications:
  - Section 3/4.15.A, "Inservice Inspection and Testing - Inservice Inspection," and Basis
  - Section 3/4.6.H.2, "Primary System Boundary - Snubbers," and Basis

- Procedures:
  - 4AWI-09.04.03, Revision 0 and Revision 1, "ASME Section XI Repair/Replacement Program"
  - 4AWI-10.01.03, Revision 6 and Revision 7, "Inservice Testing Program"
  - 4903, Revision 12, "Snubber Changeout Procedure"
  - 3186-G-01-01, Revision 4, "Quality Control Inspection"
  - ISI-VT-2.0, "Visual Examination of Components and Their Supports"

- Condition Reports:
  - 20010344, "NIS-2 Forms Not Filled out in Accordance with 1986 ASME Section XI Requirements for Snubber Replacements"
  - 20010059, "Prior to Installation of Work Incorrect LCO for Replacement of SS-707 Identified in PM-4903 as Required by AWI-02.03.03"

- Licensee corrective action plan to regain compliance with Section XI requirements for snubbers and associated flow chart.

- Work Orders:
  - 0004852, "PM 4903 (Snubber Changeout)"
  - 0004853, "PM 4903 (Replace Snubber)"
  - 0105797, "Changeout Snubber"

- Monticello, "Inservice Inspection Examination Plan Revision 3," Third Interval June 1, 1992, Through May 31, 2002

b. <u>Findings</u>

The inspectors identified one Green finding and three non-cited violations associated with this issue. The details are documented below.

On January 9, 2001, the inspectors observed Work Order 0105797, "Changeout Snubber," while performing a verification of post maintenance testing (Section 1R19). During the review of the testing, the inspectors noted that a snubber in the High Pressure Core Injection (HPCI) system was being periodically replaced. The licensee had increased the replacement frequency for this snubber in response to a reduced service life attributed to excessive wear caused by vibration of the steam inlet piping normally present during routine HPCI system operations.

The inspectors questioned the licensee about the suitability of the replacement snubber. The inspectors noted that the Code, Subsection IWA-7220, "Verification of Acceptability," requires a replacement to be evaluated for suitability for the intended service environment. The licensee was unaware of this requirement; therefore, the inspectors inquired about the opinion of the Authorized Nuclear Inservice Inspector (ANII) relative to this issue. Subsequently, the licensee indicated that the ANII believed that the snubber should be replaced with one that was not prone to vibration induced failure.

On January 17, 2001, after further review, the licensee identified that the snubber in question had been evaluated at the time it had first failed in 1993. The licensee produced an engineering evaluation on January 17 which provided additional margin. The evaluation indicated that margin in excess of the tested values was available, thereby concluding that the snubber had always remained operable. Additionally, the licensee indicated that they had discussed this information with the ANII, and that the ANII had agreed that replacement of the snubber was acceptable because no failure was identified.

Due to the differences between the two ANII assessments and because the snubber change-out was a Code replacement, the inspectors indicated that they would like to review the replacement approval documentation. The licensee's staff members involved in the discussion indicated that they were unfamiliar with all of the documentation requirements. The inspectors noted that the ANII was required to sign a Form NIS-2, "Owners Report for Repairs or Replacements," for all Code repairs or replacements. On January 19, 2001, the licensee determined that ANII involvement during repairs or replacements, and associated NIS-2 forms for related work, had not been completed for snubber maintenance. Additionally, the licensee concluded that compliance with Code requirements possibly extended to other systems or processes. Technical Specifications, Section 3.15 requires that in order to consider a Code component operable, the component must be in conformance with the Code.

However, the licensee did not declare all safety related snubbers inoperable on January 19, 2001. In response to numerous discussions between the resident inspectors and the licensee regarding compliance with the technical specifications, a call was held with between regional management and the plant manager on January 24, 2001. Subsequent to the conference call on January 24, 2001, at 12:30 p.m., the licensee entered a 72-hour allowed outage time associated with Technical Specification 3.6.H.2. On January 25, subsequent to a conference call between regional management and the licensee, the licensee determined at approximately 5:00 p.m. that they had failed to enter Technical Specification 3.6.H.2 on January 19, when recognition of the non-compliance was first noted. Based upon the actual time of discovery, the licensee determined that they had exceeded the 72-hour allowed outage time, declared the snubbers and associated equipment inoperable, and commenced a plant shutdown as required by Technical Specifications.

In parallel with the shutdown, the licensee performed reviews of inoperable snubbers to "determine through engineering evaluations that the as-found condition of the snubber had no adverse effect on the supported components and that they would retain their structural integrity in the event of a design basis seismic event." The evaluations performed allowed the licensee to comply with Technical Specification 3.6.H.2.b. The engineering evaluation for affected snubbers was completed and the plant shutdown terminated approximately 3½ hours after power reduction was commenced.

The inspectors determined that this issue was more than minor because the failure to involve the ANII in repair and replacement activities deprived the process of a third party review of the technical and quality requirements of the Code. Additionally, if left uncorrected, the lack of a third party review could become a more significant safety concern because the third party review provided additional margin to ensure that systems were maintained as originally designed. Therefore, this issue had a credible impact on safety. The inspectors also determined that with inoperable snubbers, this issue affected associated system operability, availability, reliability, and concurrently influenced mitigating systems and the seismic external event initiator. This determination was supported by the fact that snubber operability is required to ensure ECCS system operability, and that snubbers are necessary to mitigate the effects of a seismic event. This, coupled with the fact that multiple mitigating systems were potentially affected by the condition, resulted in the inspectors performing a phase one significance determination. The inspectors reviewed the impact of the issue with respect to the significance determination process (SDP) for mitigating systems and determined that the issue had an impact on operability. However, based upon the licensee's evaluation of functionality, the issue did not constitute an actual loss of safety function and the issue did not screen as risk significant with respect to seismic, fire, flooding, or severe weather initiating events, therefore, the issue was determined to be Green.

Technical Specification 3.15.A. requires that Quality Group A, B, and C (ASME Code Class 1, 2, and 3) components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code to be considered operable. Technical Specification 3.6.H.2, states, in part, "When one or more snubbers made or found to be inoperable for any reason when Operability is required, within 72 hours, [absent of meeting provisions outlined in part a or part b], c. Declare the supported system inoperable and take the action required by the Technical Specification for inoperability of that system." Contrary to the requirements of Technical Specification 3.6.H.2.c, on January 19 the licensee determined that all Code Class 1, 2, and 3 snubbers did not meet the requirements of Section XI of the Code because the ANII was not involved in repair/replacement activities through the use of the required NIS-2 forms. However, systems with inoperable snubbers were not declared inoperable within 72 hours from the time of discovery, nor was the action taken as required by the Technical Specification for the inoperable systems. This violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (50/263-01-02-01). The licensee has entered this issue into their corrective action program as Condition Report 20010344.

The Code of Federal Regulations, Title 10, Part 50.72, required that licensee's shall notify the NRC as soon as practical and in all cases with in hour of "Any event or condition during operation that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or results in the nuclear power plant being: (B) In a condition that is outside the design basis of the plant. [Part 50.72(b)(1)(ii)(B)]" The licensee's Safety Analysis Report, Section 1.2.1.a, "Principal Design Criteria - General Criteria," states, "The plant is designed, fabricated, erected, and operated to produce electrical power in a safe, reliable and efficient manner and in accordance with applicable codes and regulations." The inspectors determined that on January 19, 2001, when the licensee failed to enter the applicable 72-hour Technical Specification Limiting Condition for Operation due to Code noncompliance, the plant was being operated outside its design basis in that "applicable codes and regulations" were not being followed with respect to snubber replacements. This violation was considered more than minor because the issue had the potential to impact the NRC's ability to perform its regulatory function in responding to an event. Contrary to the requirements of 10 CFR 50.72(b)(1)(ii)(B), the licensee failed to make the appropriate notification of this condition within the required time period. This violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (50/263-01-02-02). The licensee has entered this issue into the corrective action program as Condition Report 20010801.

The inspectors also reviewed procedures 4AWI-09.04.03, Revision 0 and Revision 1, "ASME Section XI Repair/Replacement Program." The inspectors found that both revisions, dated as early as June 1997, contained requirements for repairs and replacements to be governed in accordance with Code and that the "ANII should concur with the repair/replacement plan prior to implementation." The inspectors also identified that training had been conducted for all engineering and technical staff personnel with respect to Code requirements. Additionally, the training contained material to support the requirement that repairs and replacements required NIS-2 forms. The licensee reviewed their training processes to determine why Code non-compliance was not identified during the training process or the non-compliance continued after the training was conducted. The licensee has entered this issue into their corrective action program as part of the root cause analysis for Condition Report 20010344.

Technical Specifications 6.5 requires detailed written procedures for, "surveillance and testing requirements that could have an impact on nuclear safety," and for maintenance and test procedures that satisfy routine inspection for "preventative or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety." Furthermore, the requirements established in the procedure 4AWI-09.04.03 concerning ANII concurrence with repair/replacement plans supports the surveillance requirements of Technical Specification 3/4.15. This violation was considered more than minor because the failure to follow procedures for surveillance and maintenance activities for safety related systems has the potential to have a credible impact on safety. Contrary to the above, the licensee, for the past several years, has failed to obtain ANII concurrence on repair and replacement plans prior to implementation. This violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (50/263-01-02-03). The licensee has entered this issue into their corrective action program as Condition Report 20010504.

Additionally, the licensee has determined that the condition extended to multiple systems and processes. As a result, the licensee has applied for, and been granted, a NOED (Section 1R14.3). The licensee is currently developing a plan to restore compliance with Code. The licensee has anticipated that the restoration of affected systems and components to full Code compliance will take, at a minimum, several months.

.3 Initiation of Technical Specification Required Shutdown Due to ASME Section XI Non-Compliance with Safety-Relief Valves and Associated Notice of Enforcement Discretion

a. Inspection Scope

The inspectors reviewed personnel performance, recovery actions, and licensee response to the initiation of a plant shutdown required by Technical Specifications due to non-compliance with the Code requirements for reactor pressure vessel safety relief valves (SRVs). The inspectors also reviewed the licensee's application for a NOED. To evaluate the occurrence the inspectors reviewed operator logs, equipment records, licensee response, applicability to the significance determination process, and contingency plans. Documents reviewed included:

- Technical Specifications:
  - Section 3/4.15.A, "Inservice Inspection and Testing - Inservice Inspection," and Basis
  - Section 3/4.6.H.2, "Primary System Boundary - Snubbers," and Basis

- Procedures:
  - Administrative Work Instruction 4AWI-09.04.03, Revision 1, "ASME Section XI Repair/Replacement Program"
  - Preventative Maintenance Procedure 4208-03-PM, Revision 14, "SRV (Pilot & 2nd Stage) Pilot Valve Assembly Inspections, Refurbishment, and Steam Testing
  - Preventative Maintenance Procedure 4208-PM, Revision 21, "SRV Pilot Valve Assembly (Pilot & 2nd Stage) Change-out"
  - C.3, Revision 22, "Shutdown Procedure"

- Condition Reports:
  - 20010344, "NIS-2 Forms Not Filled out in Accordance with 1986 ASME Section XI Requirements for Snubber Replacements"
  - 20010059, "Prior to Installation of Work Incorrect LCO for Replacement of SS-707 Identified in PM-4903 as Required by AWI-02.03.03"

- Licensee corrective action plan to regain compliance with Section XI requirements for snubbers and associated flow chart.

- Work Orders:
  - 0004852, "PM 4903 (Snubber Changeout)"
  - 0004853, "PM 4903 (Replace Snubber)"
  - 0105797, "Changeout Snubber"

- Monticello "Inservice Inspection Examination Plan," Revision 3, Third Interval June 1, 1992, through May 31, 2002

b. Issues and Findings

The inspectors identified one Green finding and one unresolved item associated with this issue. The details are documented below.

On January 29, 2001, during a review of the extent of condition for Code non-compliance issues (Section 1R14.2), the licensee identified that replacement of SRV actuator assemblies were not controlled in accordance with Code requirements. The licensee had determined that ANII involvement and completion of required NIS-2 forms had not been accomplished for replacement of SRV actuator assemblies. This resulted in the licensee declaring all 8 SRVs inoperable and commenced a power reduction as required by Technical Specification 3.6.E.1.

Technical Specification 3.15.A.1 states, "To be considered operable quality group A, B, and C components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and Applicable Addenda for continued service of ASME Code Class 1, 2, and 3 components, respectively, except where relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i)." The licensee concluded that application for a NOED relative to this Technical Specification, and submittal of an exigent Technical Specification amendment to move Inservice Inspection requirements to a licensee controlled document, would provide the ability to disposition the SRV non-compliance, as well as any future non-compliance issues, using the corrective action program and the Generic Letter 91-18 operability determination process.

The licensee requested enforcement discretion from the requirements of Technical Specification 3.15.A.1 until an exigent Technical Specification amendment could be processed. Concurrently with the request for a NOED, the licensee performed operability determinations to demonstrate functionality of the SRVs. The issues were addressed by the licensee in an application for a NOED that was completed in parallel with the unit shutdown activities. The licensee discontinued plant shutdown activities after being granted a NOED and completion of operability determinations for the SRVs that demonstrated that the valves remained operable but degraded with respect to Code compliance issues.

After the licensee identified that SRVs were not in Code compliance, the inspectors determined that the issue was more than minor because the failure to involve the ANII in repair and replacement activities deprived the process of a third party review of the technical and quality requirements of the Code. Additionally, if left uncorrected, the lack of a third party review could become a more significant safety concern because the third party review provided additional margin to ensure that systems were maintained as originally designed.

The inspectors also determined that inoperable SRVs could affect associated system operability, availability, and reliability. This determination was supported by the fact that SRV operability is required to ensure ECCS system operability for intermediate size loss of coolant accidents. Additionally, SRVs are necessary to mitigate multiple transients as identified in the licensee's emergency operating procedures. This resulted in the inspectors performing a phase one significance determination.

The inspectors reviewed the impact of the issue with respect to the SDP for mitigating systems and found that: the issue potentially impacted operability; based upon the licensee's evaluation of functionality, the issue did not constitute an actual loss of safety function; and the issue did not screen as risk significant with respect to external events. The issue was determined to be within the licensee response band (Green). The licensee entered this issue into their corrective action program as Condition Report 20010504.

The inspectors reviewed the circumstances that led to the need for a NOED, and the extent of the condition with respect to Code compliance, and determined that the licensee had not been accomplishing Code programs as required. Section 50.55a(a)(2) states, in part, "Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (e), (f), and (g) of this section." The licensee identified that they did not implement the Code for multiple systems and programs. This issue is being treated as a unresolved item (50-263/01-02-04) consistent with the guidance provided in NRC inspection manual chapter 9900, "Notices of Enforcement Discretion." The licensee has entered this issue into their corrective action program as Condition Report 20010344. The licensee continues to evaluate the inservice test program against the requirements of Section XI of the ASME code.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the technical adequacy of the following operability evaluations to determine the impact on Technical Specifications, and the significance of the evaluations, and to ensure that adequate justifications were documented.

- High Pressure Coolant Injection (HPCI) Snubber Replacement

- Scram Discharge Volume (SDV)

Operability evaluations were selected based upon the relationship of the safety-related system, structure, or component to risk. The documents reviewed included:

- Procedures and Forms:
  - 4903, Revision 12, "Snubber Changeout Procedure"
  - 3186-G-01-01, Revision 4, "Quality Control Inspection"
  - ISI-VT-2.0, "Visual Examination of Components and Their Supports"
  - 0006, Revision 18, "Scram Discharge Volume Hi Level Scram Test and Calibration"
  - 4AWI [Administrative Work Instruction] - 04.04.02, Revision 5, "Equipment Positioning, Witness Check, and Independent Verification Methods"

- Calculations:
  - CA-97-085, Revision 0, "Scram Discharge Volume Calculation Levels for Tech Spec Limits"

- CA-97-093, Revision 0, "Magnetrol Scram Discharge Volume Setpoint Calculation"
- CA-97-094, Revision 0, "FCI [Fluid components, Inc.] Scram Discharge Volume Setpoint Calculation"

- Condition Reports:
  - 20010059, "Prior to Installation of Work Incorrect LCO for Replacement of SS-707 Identified in PM-4903 as Required by AWI-02.03.03"
  - 20010194, "Some I&C Procedures Perform All Independent Verifications at the End of the Test for All Instruments Worked on Inconsistent With 4AWI - 04.04.02"

- P&IDs:
  - M-118, Revision AU, "Control Rod Hydraulic System"
  - M-119, Revision S, "Control Rod Hydraulic System"

- Work Orders:
  - 0004852, "PM 4903 (Snubber Changeout)"
  - 0004853, "PM 4903 (Replace Snubber)"
  - 0105797, "Changeout Snubber"

- Operations Manual Section B.1.3, "Control Rod Drive (CRD) Hydraulic System"

- USAR, Revision 18, Section 3.5.3.3, "CRD Hydraulic System"

- Technical Specifications:
  - Section 3/4.6.H, "Primary System Boundary - Snubbers," and Basis
  - Section 3/4.15.A, "Inservice Inspection and Testing - Inservice Inspection," and Basis
  - Section 3/4.1, "Reactor Protection System," and Basis

b. <u>Issues and Findings</u>

There were no findings identified during this inspection.

1R19 <u>Post-Maintenance Testing</u>

a. <u>Inspection Scope</u>

The inspectors selected the following post-maintenance activities for review. Activities were selected based upon the structure, system, or component's ability to impact risk.

- Adjust Packing "B" Feedwater Regulating Valve

- HPCI Snubber Replacement

- No. 13 Instrument Air Compressor Maintenance Work

- Control Rod Select Switch Work

The inspectors observed the performance of post-maintenance testing activities which included, but were not limited to, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, system restoration, and evaluation of test data. The inspectors verified that maintenance and post-maintenance testing activities were adequate and would detect deficiencies prior to returning equipment to service. The documents reviewed included:

- May 27, 1986, Memorandum for: Charles E. Norelius, Director, Division Reactor Projects, RIII, From: Harold R. Denton, Director, Office of Nuclear Rector Regulation, Subject: "Technical Specification Interpretation On Snubbers"

- July 9, 1999, Letter from Frank Rinaldi, Project Directorate II to Mr. H. B. Barron, Duke Energy Corporation, Subject: "McGuire Nuclear Station, Units 1 and 2, RE: Licensing Position Regarding Snubbers (TAC Nos. MA5519 and MA5520)"

- Operations Manual, Section B.5.5, "Reactor Manual Control System"

- USAR, Revision 18, Section 7.2.1, "Reactor Manual Control System"

- Technical Specifications:
  - Section 3/4.6.H, "Primary System Boundary - Snubbers," and Basis
  - Section 3/4.5, "Core and Containment Spray/Cooling Systems," and Basis
  - Section 3/4.15.A, "Inservice Inspection and Testing - Inservice Inspection," and Basis

- Work Orders:
  - 0004429, "Moderate Packing Leak on CV-6-12B"
  - 0004852, "PM 4903 (Snubber Changeout)"
  - 0004853, "PM 4903 (Replace Snubber)"
  - 0105797, "Changeout Snubber"
  - 0003224, "Reactor Manual Control Rod Select Matrix Switch Replacement"
  - 9905451, "Replace TD-A and TD-C in Instrument Air Panel # 2"
  - 0105949, "Adjust 11 and 13 Air Compressor Time Delay Relays"
  - 0105944, "Investigate 13 Air Compressor Unloader Controls"

- Condition Report 20010059, "Prior to Installation of Work Incorrect LCO for Replacement of SS-707 Identified in PM-4903 as Required by AWI-02.03.03"

- Procedures and Forms:
  - 4903, Revision 12, "Snubber Changeout Procedure"
  - 3186-G-01-01, Revision 4, "Quality Control Inspection"
  - ISI-VT-2.0, "Visual Examination of Components and Their Supports"
  - 3069, Revision 8, "Post-Maintenance Activities Control Cover Sheet"
  - 3560, "Revision 5, "Infrequent Test or Evolution Briefing Guide"
  - 0074, Revision 27, "Control Rod Drive Exercise"

b. Issues and Findings

There were no findings identified during this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors selected SDV Instrument Calibration Checks for review. This activity was selected based upon risk significance and the impact upon risk that an unidentified performance degradation of a structure, system, or component could have if unresolved for long periods of time.

The inspectors observed the performance of the surveillance testing activities, including reviews for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, impact of testing relative to performance indicator reporting, and evaluation of test data. The following documents were reviewed:

- Drawings:
  - M-118, Revision AU, "Control Rod Hydraulic System"
  - M-119, Revision S, "Control Rod Hydraulic System"

- Surveillance Test Procedure 0006, Revision 18, "Scram Discharge Volume Hi Level Scram Test and Calibration"

- Calculations:
  - CA-97-085, Revision 0, "Scram Discharge Volume Calculation Levels for Tech Spec Limits"
  - CA-97-093, Revision 0, "Magnetrol Scram Discharge Volume Setpoint Calculation"
  - CA-97-094, Revision 0, "FCI [Fluid components, Inc.] Scram Discharge Volume Setpoint Calculation"

- Operations Manual Section B.1.3, "Control Rod Drive (CRD) Hydraulic System"

- Technical Specifications Section 3/4.1, "Reactor Protection System," and Basis

- USAR, Revision 18, Section 3.5.3.3, "CRD Hydraulic System"

- Radiation Work Permit (RWP) 26, Revision 10, "Miscellaneous Post Contaminated Instrument Racks"

- Licensee Event Report (LER) 50-263/1986-025, "Scram During SDV Surveillance Test"

- 4AWI - 04.04.02, Revision 5, "Equipment Positioning, Witness Check, and Independent Verification Methods"

b. Issues and Findings

There were no findings identified during this inspection.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

On August 29, 2000, the licensee submitted, by letter, Revision 19 to the Monticello Nuclear Generating Plant Emergency Plan. The inspector reviewed the submittal in order to determine whether the changes in Revision 19 might decrease the emergency plan's effectiveness pending future inspection of the implementation of these changes. This emergency plan revision was submitted in accordance with 10 CFR 50.54(q).

b. Observation and Findings

There were no findings identified during this inspection.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (IP 71151)

Cornerstone: Mitigating Systems

Safety System Functional Failures

a. Inspection Scope

The inspectors verified the accuracy and completeness of the "Safety System Unavailability" performance indicator data submitted by the licensee for January 1 through June 30, 2000, to monitor the readiness of important safety systems to perform their safety functions in response to off-normal events or accidents. The inspectors reviewed Emergency Diesel Generator (EDG), Residual Heat Removal (RHR), and Reactor Core Isolation (RCIC) system data reported to the NRC since the last verification. The review was accomplished, in part, through evaluation of the Technical Specification requirements, plant records, procedural reviews, and reactor coolant sample data. The procedures evaluated and documents reviewed included:

- Monticello Performance Indicator Data Summary Report Q3/2000
- Nuclear Energy Institute 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline"
- 4AWI-04.08.11, Revision 1, "NRC Performance Indicator Reporting"
- Monticello Operations Daily Log - Part J, Revision 76

- Worksheet 3530-05, Revision 0, "Performance Indicator Safety System Unavailability," April 2000 - June 2000

- Worksheet 3530-10, Revision 0, "Performance Indicator Mitigating Systems for 11 EDG, 12 EDG, RCIC, A Loop RHR, and B Loop RHR," April 2000 - June 2000

- Maintenance Rule Summary "Availability/Reliability RHR System, December 1999 - December 2000"

- Maintenance Rule System Performance Data:
  - RHR System, January 2000 - June 2000
  - RCIC System, January 2000 - June 2000
  - Emergency Diesel Generators, January 2000 - June 2000

b. Issues and Findings

There were no findings identified during this inspection.

4OA3 Event Follow-up

Cornerstone: Mitigating Systems, Barrier Integrity, Emergency Preparedness

.1 (Closed) LER 50-263/2000-012: Inoperable Containment Isolation Valve Results in a Condition Prohibited by Technical Specifications.

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-012, "Inoperable Containment Isolation Valve Results in a Condition Prohibited by Technical Specifications." The inspectors reviewed the following references:

- Condition Report 20003281, "MO-2373 Failure to Indicate Fully Open Upon Initiation from the Control Room"

- Technical Specification, Section 3/4.7, "Containment Systems," and Basis

- USAR, Revision 18, Section 5.2, "Primary Containment System"

- Operations Manual Section B.4.1, "Primary Containment"

b. Issues and Findings

On August 27, 2000, while shutting down the plant to effect repairs on a transformer, the inboard main steam line drain containment isolation valve, MO-2373, failed to indicate fully open following manual actuation from the control room. The licensee subsequently investigated this condition and determined that MO-2373 had been inoperable since the valve had been maintained during the 1998 refueling outage, and that the redundant containment isolation valve, MO-2374, was not closed as required by Technical

Specifications 3.7.A.2.a.(1) and 3.7.D.2. The inspectors reviewed the licensee's assessment that the event had no effect on the health and safety of the public, and concluded that the failure to close the redundant containment isolation valve, MO-2374, as required by Technical Specifications constituted a violation of minor significance that was not subject to enforcement action in accordance with Section IV of the Enforcement Policy. The licensee had entered this issue into their corrective action program as Condition Report 20003281.

.2 (Closed) LER 50-263/2000-013: Initiation of Containment Purge Prior to Sampling and Analysis of Containment Atmosphere in Violation of Technical Specifications.

a. Inspection Scope

The inspectors evaluated LER 50-263/2000-013, "Initiation of Containment Purge Prior to Sampling and Analysis of Containment Atmosphere in Violation of Technical Specifications," Revision 0 and Revision 1. The inspectors reviewed the following references:

- Condition Report 20003350, "Primary Containment Purging for Commencement of Inerting Initiated Prior to Sampling and Analysis Completion"

- Technical Specification, Section 3/4.8, "Radioactive Effluents," and Basis

b. Issues and Findings

On the morning of September 2, 2000, the licensee was in the process of unit restart following a brief maintenance outage. Upon review of the prestart checklist, the licensee identified that no containment atmosphere grab sample had been obtained prior to the initiation of containment purging with nitrogen, as required by Technical Specification 4.8.B.6.b. The inspectors reviewed the licensee's conclusion that the event had no effect on the health and safety of the public, and determined that the failure to obtain and analyze a containment atmosphere grab sample prior to initiation of the containment purge with nitrogen as required by Technical Specifications constituted a violation of minor significance that was not subject to enforcement action in accordance with Section IV of the Enforcement Policy. The licensee had entered this issue into their corrective action program as Condition Report 20003350.

4OA6 Meetings, including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Morris and other members of licensee management on February 14, 2001. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>

J. Morris, Site General Manager
B. Day, Plant Manager
J. Grubb, General Superintendent, Engineering
K. Jepson, Superintendent, Chemistry and Environmental Protection
B. Linde, Superintendent, Security
B. Sawatzke, General Superintendent, Maintenance
C. Schibonski, General Superintendent, Safety Assessment
E. Sopkin, General Superintendent, Operations
L. Wilkerson, Manager, Quality Services
J. Windschill, General Superintendent, Radiation Services
D. Neve, Acting Licensing Project Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened & Closed

50-263/01-02-01	NCV	Inoperable snubbers not declared inoperable as required by Technical Specifications (Section 1R14.2)
50-263/01-02-02	NCV	Failure to make notifications as required by 10 CFR 50.72 (Section 1R14.2)
50-263/01-02-03	NCV	Failure to follow established procedures for surveillance and testing as required by Technical Specifications (Section 1R14.2)
50-263/01-02-04	URI	The requirements of the ASME Boiler & Pressure Vessel Code, Section XI, as required by 10CFR50.55a (Section 1R14.3)

Closed

50-263/2000-012	LER	Inoperable Containment Isolation Valve Results in a Condition Prohibited by Technical Specifications (4OA3)
50-263/2000-013	LER	Initiation of Containment Purge Prior to Sampling and Analysis of Containment Atmosphere in Violation of Technical Specifications (4OA3)

Discussed

None

LIST OF ACRONYMS USED

ANII	Authorized Nuclear Inservice Inspector
ARP	Alarm Response Procedure
AWI	Administrative Work Instruction
CAM	Continuous Air Monitor
CFM	Cubic Feet per Minute
CRD	Control Rod Drive
CRV	Control Room Ventilation
DBD	Design Basis Document
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NOED	Notice of Enforcement Discretion
NUMARC	Nuclear Management and Resources Council
P&ID	Piping and Instrument Diagram
PRM	Process Radiation Monitor
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RWP	Radiation Work Permit
SCT	Secondary Containment
SDP	Significance Determination Process
SDV	Scram Discharge Volume
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SRV	Safety Relief Valve
USAR	Updated Safety Analysis Report
WRGM	Wide Range Gas Monitor